Worcester Polytechnic Institute
Digital WPI

Major Qualifying Projects (All Years)

Major Qualifying Projects

April 2016

Investigating High Temperature Gas-Cooled Reactors for Research Applications

Emilia Monika Konert
Worcester Polytechnic Institute

Investigating High Temperature Gas-Cooled Reactors for Research Applications

EMILIA KONERT

WORCESTER POLYTECHNIC INSTITUTE

MAJOR QUALIFYING PROJECT

PHYSICS, APRIL 2016

Advisor: Professor David Medich

ABSTRACT

Using existing research, analytical methods, and computational simulations, the feasibility of a 10 MW HTGR for research purposes is explored. Great strides are underway in fission technology for energy production, and few studies have been conducted on HTGRs for research applications. These reactors are compact, have long refueling periods, and inherent safety features. Valuing safety, cost, and ease of operation, this study proposes a HTGR pebble-bed reactor (PBR) be designed for university research, and sets up a framework for future projects at WPI.

I. INTRODUCTION

The premise of this study is based upon HTGRs for research purposes, but their function as energy production cannot be dismissed. As the world develops and fossil fuel supply rapidly decreases, alternative non-destructive forms of energy are required. Energy and research reactors can be developed concurrently, due to the modular quality, low power, and high flux of Generation IV fission reactors. This paper examines the benefits and characteristics of HTGRs, uses analytical methods to further investigate a simple reactor design for research purposes, and sets up a framework for computational methods using Monte-Carlo N-Particle Transport Code (MCNP). Ultimately, this study was the first step in designing the first Generation IV HTGR university reactor in the United States.

1.2 Background

HTGRs are a type of small modular reactor (SMR), which are classified as generation IV reactors. Generation IV reactors are highly economical, posses inherent safety features, produce minimal waste, and are proliferation resistant. (IAEA, 2014) These are a far cry from current commercial power reactors, which require massive amounts of material, use large pools of water, and have plenty of safety concerns.

No generation IV reactors are currently active, but some projects are under construction and multitudes of designs exist. They include pressurized water reactors, molten salt reactors, liquid metal cooled reactors, and high temperature gas cooled reactors (HTGRs).
From these models, a HTGR design with a pebble bed core was chosen based on the following qualities:

i. Safety
ii. Ease of operation
iii. Cost

Safety

Due to small core size, all HTGRs have limited power levels and utilize passive features to achieve complete shutdown. Inherent safety is stressed in every HTGR design, and it is near impossible for any any harmful release of radioactive materials to occur. A low energy density in the core and large heat capacity of the shielding graphite means that the reactor will take days to heat up to a critical level – providing ample time for response in case of such a situation. The insertion of the control rods into the core is controlled by gravity, and no complicated machinery is involved, decreasing potential mechanical errors. Safety is heightened even further in pebble bed reactors by the use of fuel "pebbles" - individual spheres of fuel, in many cases uranium dioxide (UO_2) encased in graphite and ceramic, both to reflect and moderate the reaction. These fuel spheres can be heated to extreme temperatures without fuel degradation. (ORNL) This type of fuel also prevents weapons proliferation, and actually can recycle or utilize unused weapons fuels such as highly enriched plutonium. See Section 1.3 for more on pebble fuel.

Ease of Operation

Another attractive characteristic of HTGRs is their low maintenance. These reactors are designed to function with little human interaction, and refueling periods can be up to a few years. (IAEA). In pebble bed reactors, refueling is able to be conducted while the reactor is functional and on-line, by simple insertion of the pebble fuel into the core. This is not the case with reactors like current light water reactors, where the entire facility must be off-line to conduct the time consuming refueling process.

Cost

The size of these reactors also contributes to cost and construction. Because of their low power and size, these reactors naturally require less material to shield. Building costs of a reactor facility would be significantly less than those of a Generation II light water reactor, which utilizes a massive pool of water and complicated shields. Most designs are also intended to be built underground, reducing additional structural costs. In a PBR HTGR, helium (which is chemically inert) is used as a coolant. Very high flux is able to be reached in a small core, which makes HTGRs ideal for research purposes where the focus is on neutron production.

Additionally, modular reactors benefit from economics of scale – the pattern of decreased fixed cost per unit as scale of output increases. In theory, if these reactors were commercialized, this pattern would follow.

An essential factor in HTGRs becoming a reality, both as means of energy production and research reactors is public acceptance. General opinion of nuclear power has historically been very skeptical and negative, largely due to reactor failures like Chernobyl or Fukushima. Those with little knowledge of physics or reactor design will understandably be skeptical of the construction of nuclear power plants. Little has been done by energy agencies or lobbyists to market towards public opinion, which could come to be the deciding factor on HTGR reactor projects in the near future. A reactor concept like a PBR has some potential to gain favor, as it is a simple and straightforward design with new and widely unheard of technology, such as fuel spheres. The attractive core neutronics and statistic impossibility of core meltdown can easily explained and simplified for the layperson.

1.3 FUEL

The most common type of HTGR design uses pebbles as fuel. These sphere pellets of fuel are complex and a very promising technology for the future of nuclear reactors. Tristructural isotropic (TRISO) fuel particles are currently under studies in the U.S. by Idaho National Laboratory and Oak Ridge National Laboratory, both which have had great success in their safety testing. These tiny TRISO fuel particles are less than a millimeter in diameter, typically UO2 or UCO. They are coated in a layer of carbon (graphite), a thin layer of silicon carbide, then another shell of carbon. Approximately 25,000 of the particles are then combined and fabricated into a sphere (Figure 1). These larger pebbles range from 5cm to 9cm, and 7g to 11g, depending on the application and the flux desired (Reitsma) (Figure 2).

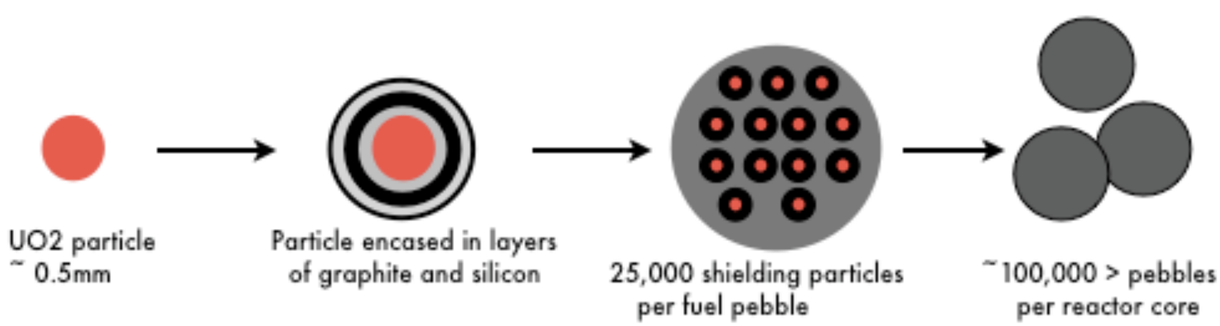


Figure 1: TRISO fuel pebble construction

TRISO fuel has been tested at INL and ORNL to observe the fuel under postulated accident conditions. The pebbles were irradiated for 600 days at about 2-3 times higher than they would be in an HTGR. No particles released fission products that could indicate failure of the coating layers. (AtomicInsights) In 2013, the irradiated pellets were inspected after being heated up to 1800 C, and only a few particles were observed to have any form of failure.

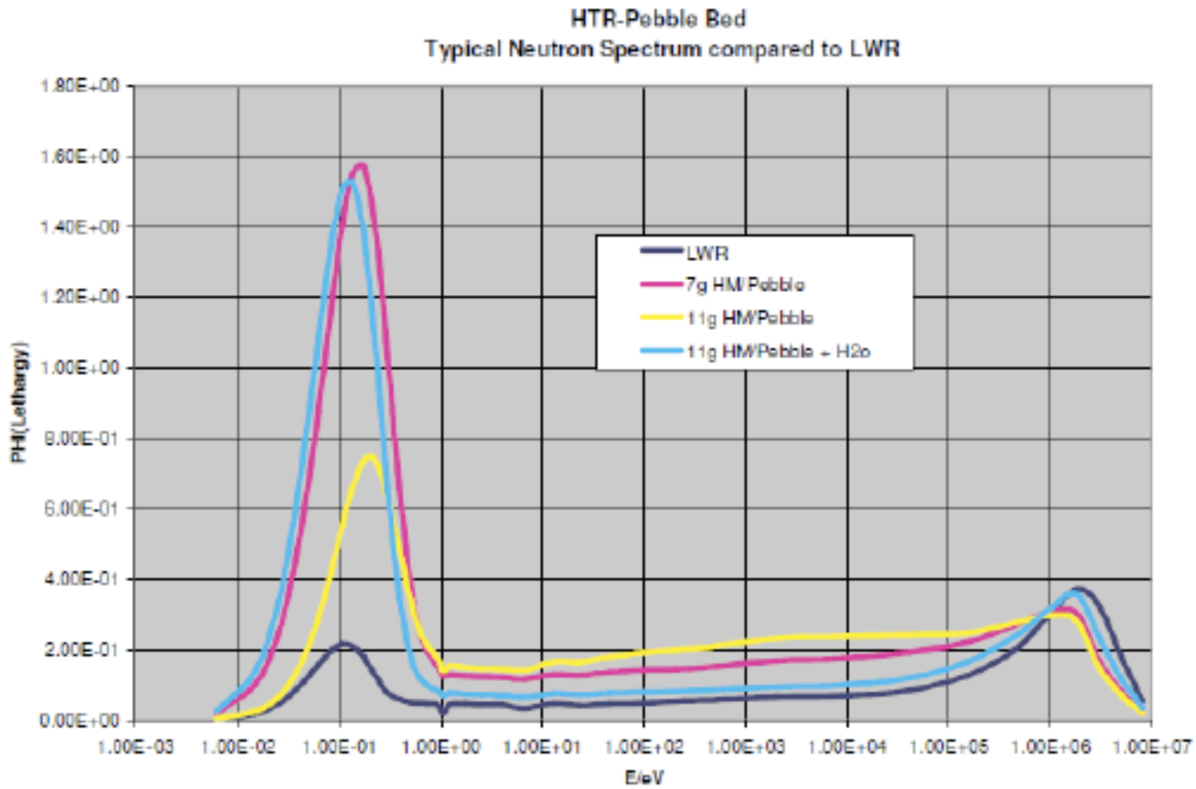


Figure 2: HTGR Pebble Bed with TRISO fuel, energy vs. lethargy. Comparison of fuel pebble weight.(Reitsma)

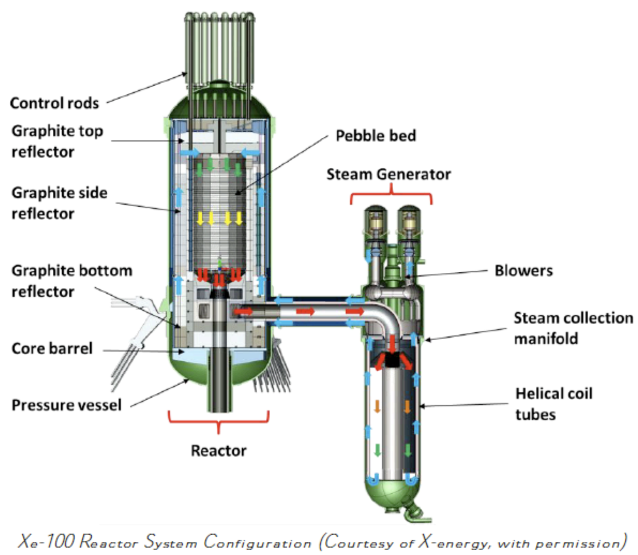
1.4 Reactor Designs

1.4.1 Energy production

No generation IV HTGRs exist yet for energy production. These technologies are relatively new, and few designs are being pursued in the United States. The following two designs are being funded and researched currently.

Xe-100

A promising concept is X-energy's Xe-100, a 100MW pebble bed reactor that is currently funded by the Department of Energy with a $40 million development grant. The design favors simplicity, modularization, and reduction of construction time. The company is working directly with ORNL and INL to advance TRISO fuel. Plant safety is promised by a power density of 3.7 MW/m^3 , the TRISO fuel with 10% enriched UCO, and underground modular construction (Figure 3). Xe-100's stability relies on nuclear, thermal, chemical and mechanical stability to prevent the core from overheating and fuel elements from cracking or changing composition (IAEA).



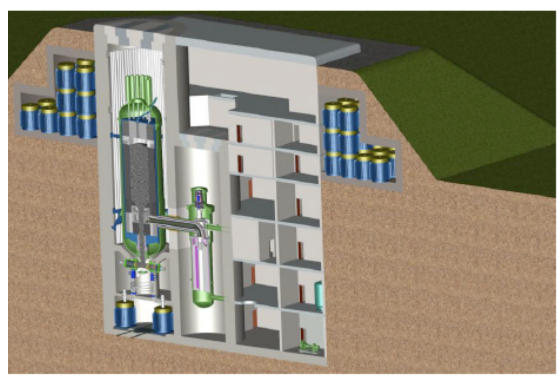


Figure 3: Xe-100 reactor concept (IAEA)

SC-HTGR

AREVA has created a design for a steam cycle HTGR with 625 MW power. Like the Xe-100, it also boats intrinsic safety characteristics and is being backed for commercialization by a industry alliance called NGNP. The design builds on past HTGR concepts and has a prismatic block type core (Figure 4) The prismatic blocks are also TRISO coated fuel particles, in block form rather than spheres.

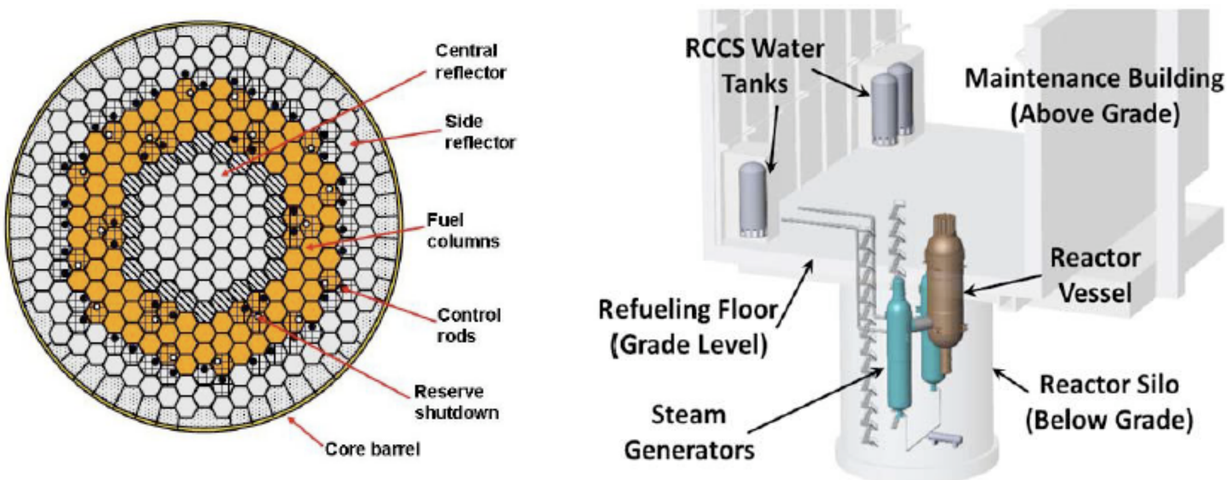


Figure 4: AREVA SC-HTGR core and reactor layout.

1.4.2 Research Reactors

For the purposes of this research, focus on reactors for neutron production is more pertinent than energy production. Research reactors act as neutron factories - creating radioisotopes for medical purposes, scattering experiments, and materials testing. One research type reactor in particular inspired the design for this study.

HTR-10

HTR-10 in China, constructed in 1995, reached first criticality in 2000. It was a 10MW (thermal) reactor that was intended to be a test for the future,higher power and modular arrangement for power production,. The project is expected to be completed by 2017. Although the project expanded into energy production, the initial 10MW design had great potential for research applications.

The HTR-10 is based on a concept first developed in Germany in the 70s, and like many new reactor models, is a reincarnation of an old technology. The fuel pebbles, using TRISO technology, are designed to withstand temperatures up to 1600 C, much higher than the normal operational temperature of the reactor, which is 700-950 C. (Zongxin) As mentioned in section 1.2 above, this design follows the passive heat transport and cooling mechanisms desired by HTGRs. Because the core has such high activity retention due to the fuel spheres, the reactor facility does not need to be airtight and can be accessed in case of accident. The core and the steam generator are separated into two pressure vessels that further prevent components of the reactor from overheating, also allowing for easier access to all parts of the reactor.

A key component of the reactor design is the passive heat removal system. Because the proposed reactor in this study will be likely built near a university campus, safety is of utmost importance and will be stressed throughout this analysis. The main heat transfer system uses a helium circulator, a steam generator and pump, steam turbine, and circulating water system as seen in Figure 5. During a normal shutdown, the helium circulator would run and lower the core decay heat. In an accident or loss of coolant, the helium circulator would not be effective. In this case, all the decay heat would dissipate through the core structure by means of conduction and radiation to outside shielding. There are coolers in the reactor cavity that use air circulation, and further cooling would be provided by natural water circulation that could continue in the system. Because the energy density of this small core is so low, even in emergency situations the decay heat could dissipate through the shield of the reactor. Because the fuel spheres are able to withstand such high temperatures and the reactor cavity can be naturally circulated with air – a rise above this temperature would not be possible. (Yuanhui)

There are two shutdown systems that can function independent of each other in the HTR-10. The first is a control rod system, the second a absorber ball system. Both are located in the graphite reflector.

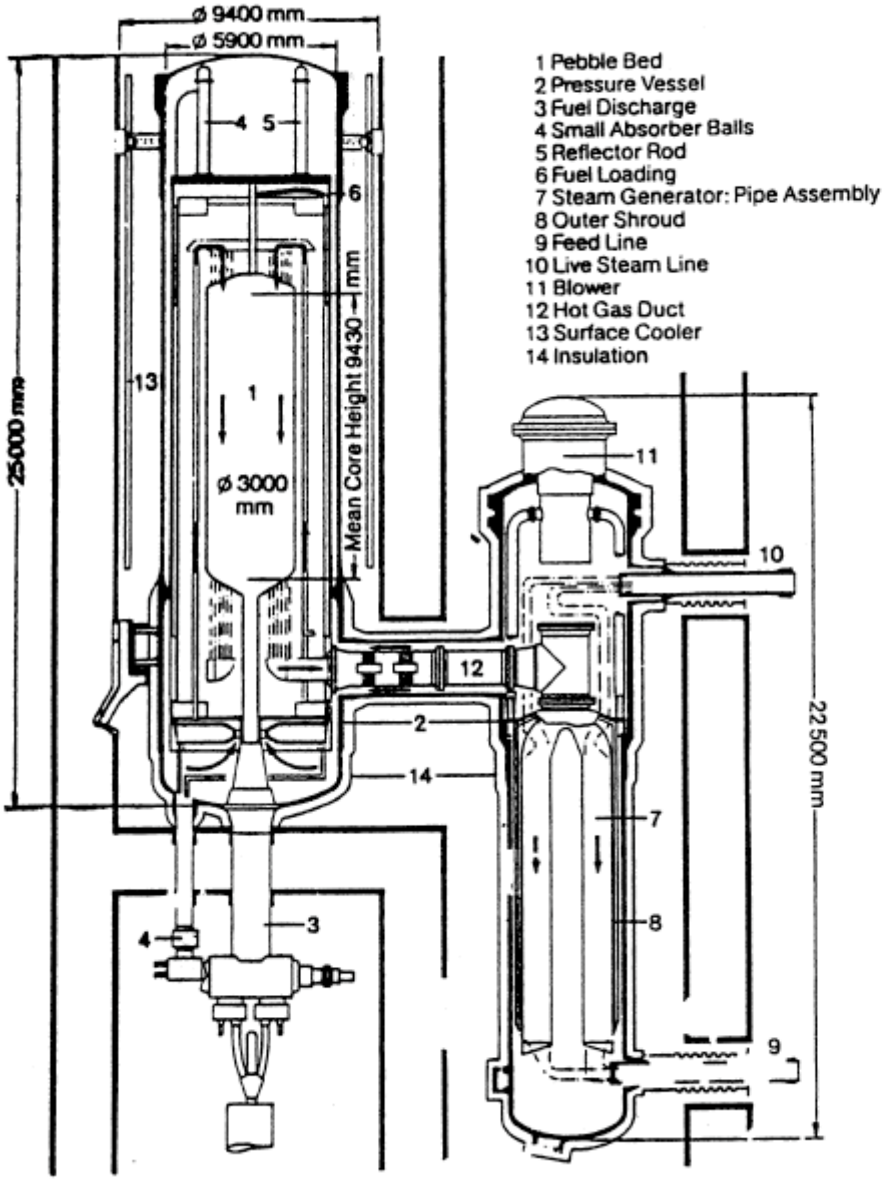


Figure 5: HTR-10 detailed design (Zongxin)

There are 10 control rods that can drop into the core on command, using a simple motor. The ball system is designed in case of the failure of the control rods. If the rods cannot drop, spheres similar in size to the fuel pebbles will drop into the core after the holding tanks are opened on demand, and fall by gravity. (Zongxin) More detailed specifications of the HTR-10 can be seen in Figure 6.

Parameters	Unit	Value
Reactor thermal power	MW	10
Active core volume	m^3	5
Average power density	MW/m^3	2
Primary helium pressure	MPa	3
Helium inlet temperature	°C	250/300
Helium outlet temperature	°C	700/900
Helium mass flow rate	kg/s	4.3/3.2
Fuel		UO_2
U-235 enrichment of fresh fuel elements	%	17
Diameter of spherical fuel elements	mm	60
Number of spherical fuel elements		27,000
Refueling mode		Multi-pass, continuously
Average discharge burnup	MWd/t	80,000

Figure 6 : HTR-10 specifications (Zongxin)

1.5 WPI Reactor

The principal purpose of this study was to determine whether a low power pebble bed HTGR would have sufficient flux for experiments and if it could compete with other university research reactors in the United States. An analytical approach was taken to estimate shielding and flux measurements. However, analytical methods cannot accurately account for neutron scatter, so the framework for Monte Carlo Method was created with MCNP to create the geometry for future calculations and to find dose rate 30 cm outside the reactor. The design of the HTR-10 was first used as inspiration to create a simplified core geometry. It was reduced to cylindrical, 10MW, 20% UO_2 enriched core with the dimensions of the HTR-10, and ordinary concrete shield.

II. METHODS

2.1 ANALYTICAL CALCULATIONS

Analytic calculations that were used focused on neutron cross sections, flux density, and attenuation of neutrons and gamma rays. These calculations are approximations, and not exact. Reactor calculations of this kind do not sufficiently take into account neutron scattering, only neutron absorption. However, scattering can be included using the Monte Carlo Method., as discussed in Section III.

2.1.1 Reactor Neutronics

When a neutron beam passes through a material, it will emerge with reduced intensity due to scattering, absorption, and other radioactive emissions (e.g. gamma emission). The beam intensity, I, is measured in terms of flux density, or neutron flux. This is equal to the number of neutrons crossing the area perpendicular to the area in 1 second. In the 10MW reactor, a simple conversion into flux can be made. This flux is an estimate, but a useful one for approximate calculations. The desired flux is on the magnitude of 10^{16} , which is considered a high flux and sufficient for research purposes. As mentioned previously, the highest flux the MIT Research reactor reaches is also on the scale of 10^{16} , and University of Massachusetts - Lowell's reactor produces ~10^{13}.

It is important to note neutron cross sections, denoted by σ, dictate the probability of scattering, absorption, and induced fission processes. The effect of all losses in intensity is the total cross section,

$$\sigma_t = \sigma_{scatter} + \sigma_{absorption}$$

In order to calculate the shielding needed for this reactor, attenuation of neutrons was investigated. The intensity of an emerging beam of neutrons after scattering and absorption with another material can be found with a simple calculation. A homogenous neutron beam passing through a material (1 cm^2 cross sectional area) has incident flux, I_o and flux I after passing through a distance x of the given medium. (Figure 7)

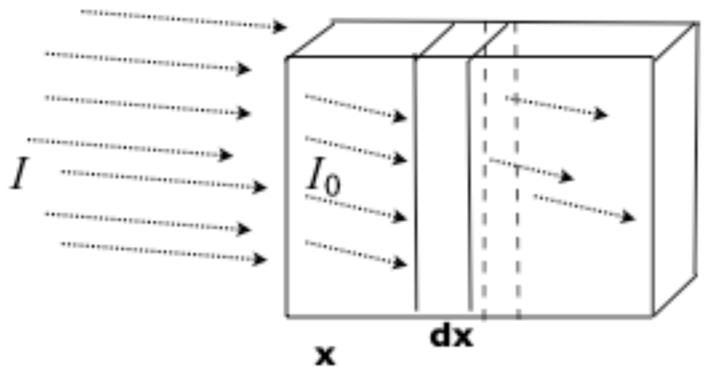


Figure 7: Attenuation of neutrons through a slab of material

Integrating for the intensity,

$$r = \sigma N_o I A$$

$$-dI = \sigma N_o I dx$$

$$\frac{dI}{I} = \sigma N_o dx = \Sigma dx$$

$$-\int_{I_0}^{I} \frac{dI}{I} = \Sigma \int_0^x dx$$

$$-log\frac{I}{I_0} = \Sigma x$$

$$\boxed{I = I_0 e^{-\Sigma x}}$$

2.1.2 Neutron Energy

Neutrons involved in nuclear reactions can be classified into 4 sections.

i. Fast neutrons: ~10 MeV to ~0.1 MeV

ii.Intermediate neutrons: ~0.1Mev to ~1000 eV

iii. Epithermal neutrons: ~1000 eV to ~ 1eV

iv. Thermal neutrons: ~1eV and less

The proposed reactor is enriched with U-235, the total fission energy vs. total cross sections can be seen in Figure 8. Here information about the energy distribution can be used for many types of calculations.

The fast neutron region is of particular interest because it has the highest energies, which correspond with more dangerous radiation levels. High flux reactors produce high numbers of

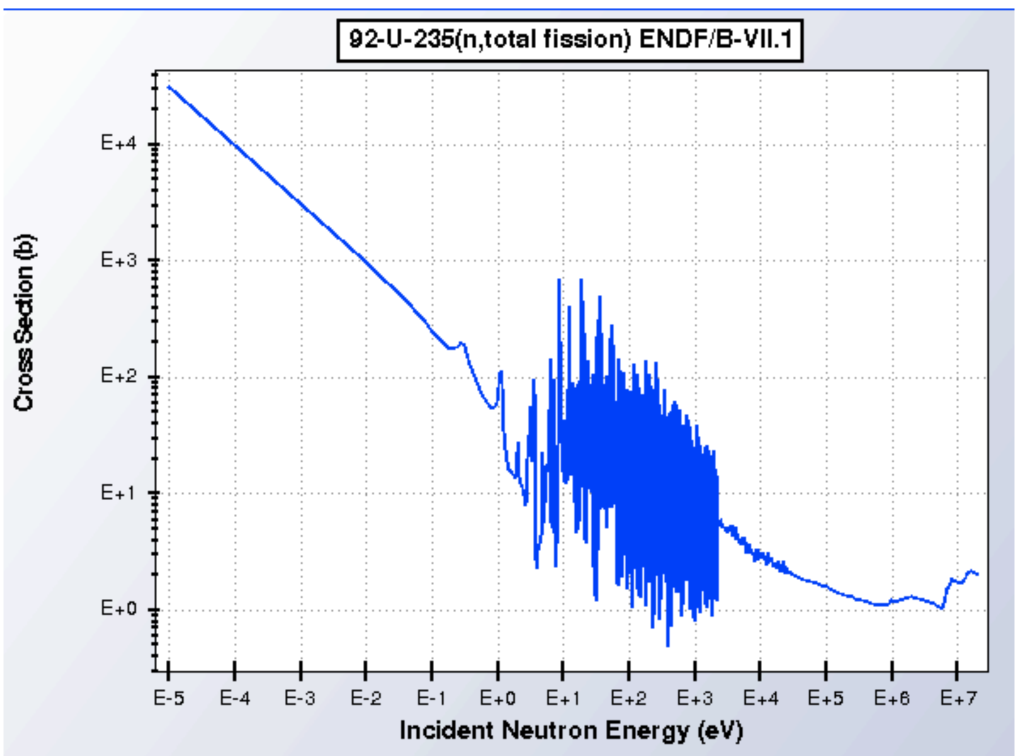


Figure 8: Total fission spectrum for U-235.
(National Nuclear Data Center)

fast neutrons. The most probable interaction between nuclei and neutrons in this energy region is for them to scatter - meaning the absorption cross section will be significantly less than the scattering cross section, $\sigma_{scatter} >> \sigma_{absorption}$.

1.2.3 Shielding Material

For the shielding material to be effective in reducing both fast neutron and gamma radiation, it requires a mixture of heavy and light nuclei to allow for all types of particle scattering to happen simultaneously. A inexpensive, low maintenance material is concrete. It contains both hydrogen and heavy elements, and can be impregnated with other elements to increase its density.

1.2.4 Absorbed Dose

Absorbed Dose is the quantity of radiation energy absorbed by matter from ionizing radiation. It is defined,

$$D = \frac{\Delta E}{m} = \frac{energy\ deposition}{mass} \quad [\mathrm{rad}\ or\ \mathrm{Gy}]$$

This value is generally expressed in gray (Gy) or rad. (1 Gy = 100 rads)
The energy deposition rate can be estimated using linear energy transfer (LET), where ϕ is the particle flux.

$$\dot{D} = \frac{\phi\, LET}{\rho}$$

Similarly to the calculation of attenuation, the computation can be visualized with flux passing through a section of medium with density ρ and can be integrated over time. Average estimations are suitable for the purpose of this study, so the average flux can be used for this calculation.

1.3 COMPUTATIONAL METHODS

The Monte Carlo method is necessary in reactor calculations, due to the heavy dependence on particle probabilities, such as neutron cross sections. It is also able to predict neutron scatter.

This paper contains only the preliminary simulation for a 100cm shield. This simulation was run first to save time and become comfortable with using and interpreting data from MCNP. An additional report was created to have a more sophisticated MCNP calculation by reducing error and creating a more complex geometry. The goal of this paper however, was to create a framework to make the next calculations less time consuming, and to verify that analytical estimations and MCNP simple calculations are in agreement.

The geometry of the reactor, as mentioned in Section 1.5, was greatly simplified from the HTR-10 model for the initial calculations. The core was reduced to a solid 90 cm x 100 cm cylinder of 20% UO^2 surrounded by ordinary concrete shielding. The outside world was air, where a sphere was placed 30cm away from the concrete wall to calculate average flux and energy deposition of the neutrons after passing through the shielding. The end of the world was defined as a cylinder 800 cm away from the origin in x, y, z. (Figure 9)

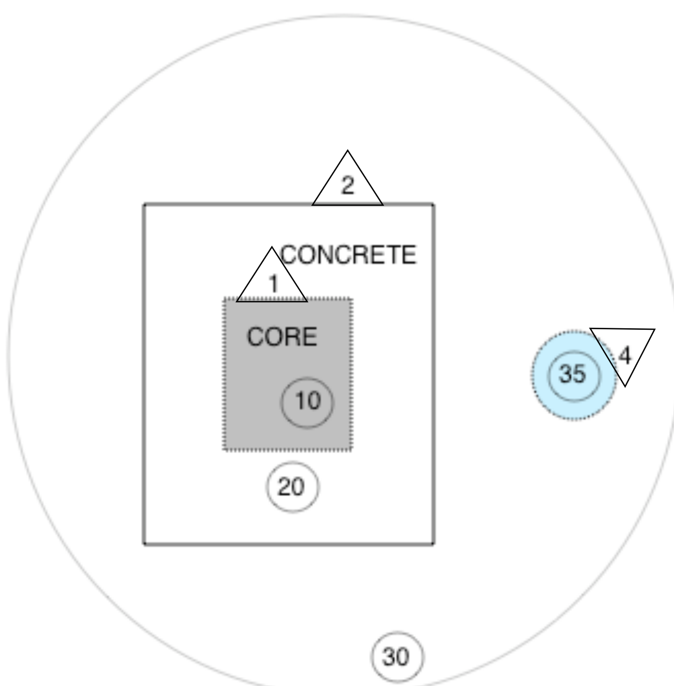


Figure 9: MCNP geometry of proposed reactor. Numbers inside circles denote cell number, and those in triangles identify their surface macrobodies.The colored sphere indicates where flux was measured.

The source was defined as 20% UO_2, rounding up the value of actual enrichment in the HTR-10 model (~17%). The fission turnoff card (NONU) was used in the initial code. This card simplifies the simulation by treating fission as simple capture. NONU is used in non-criticality mode because the fission neutrons had already been accounted for in the SDEF card.

To compare the computation to the numerical result, an F6 energy deposition tally was run. The value should theoretically be in the range of the estimated energy deposition.

The MCNP input file was as follows:

```
c HTGR Simplified model
c
c Cell cards
c
10 1 -10.97        (-1)    imp:n,n=1      $ core
20 2 -2.3         (-2 1)   imp:n,n=1      $ concrete
30 3 -0.00125     (2 4 -3) imp:n,n=1    $ outside world
35 3 -0.00125     (-4)       imp:n,n=1    $ sphere for flux
40 0              (3)        imp:n,n=0    $ end of the world

c Surface cards
c
1 RCC 0 0 0      0 100 0   45   $core
2 RCC 0 -100 0    0 300 0  145  $concrete
3 RCC 0 -800 0  0 800 0   800  $world
4 S   160 50 0            30   $sphere of air

c data cards
c core is cylinder of UO2
m1 92235.16c -0.881498 8016 -0.118502      $ UO2, avg temperature 900K
m2 1001 -0.022100  6012 -0.002484  8016 -0.05743930  11023 -0.015208
        12000  -0.001266   13027  -0.019953   14000  -0.304627   19000
-0.010045
     20000 -0.042951  26000 -0.006435      $ concrete
m3  6000  -0.000124   7014  -0.775268   8016  -0.231781   18000  -0.012827
$air
c
mode n
c Watt fission spectrum of UO2 source
SDEF POS 0 0 0  AXS 0 1 0 EXT=d1  RAD=d2  ERG=d3
```

```
SI1 0 90      $length of core
SP1 0 1       $probability
c
c
SI2 0 45     $radius of cylinder
SP2 -21 1    $ uniform distribution
c describe the watt fission spectrum
SP3 -3  1.025  2.926
c
c energy over test sphere cell
F06:n  35
c
c
nps 1000000000
```

III. RESULTS

3.1 Analytical calculations:

To find approximate neutron flux,

10MW requires 10 J of energy to be produced every second,

10 MW = 10 MJ = 6.2×10^{18} MeV ÷ 200.7 MeV/fission = 3.1×10^{16} fissions / s

Neutrons per fission: 2.4-2.8 *(IEAE)*

3.1×10^{16} fissions /s ÷ 2.4 Neutrons/ fission = 1.19×10^{16} neutrons / s

Neutron attenuation in concrete:

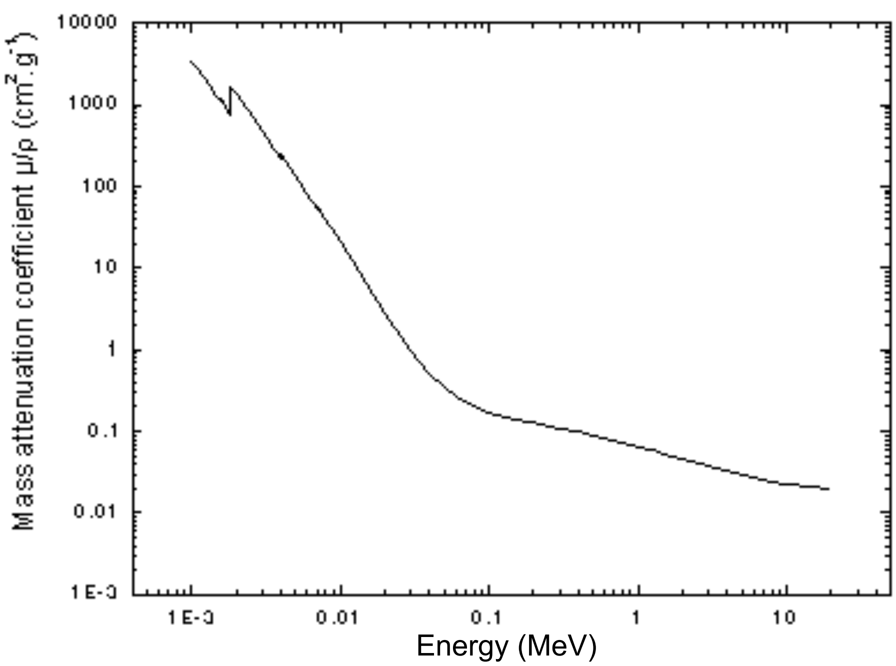


Figure 10: mass attenuation coefficient in concrete (NIST)

Using the highest neutron energy in the U-235 spectrum, ~10 MeV, the shielding required to reduce the beam to a safe flux of ~10^1 is estimated to be ~100cm.

3.2 ENERGY DEPOSITION

The absorbed dose average over the neutron energy spectrum of U-235 was calculated using

$D = \frac{\Delta E}{m}$ as discussed in the methods.

Where the elemental mass of the volume is the mass of the concrete shield, because we want to find the energy deposition outside the concrete surface. Choosing a distribution of energies from thermal to fast neutrons, an average value was calculated.

To find volume of the concrete, subtract the volume of the core cylinder from the concrete cylinder.

$V_{concrete} - V_{core} = (\ \pi\,(45cm)^2\ 100cm\ ) - (\pi\,(145cm)^2\ 300cm\ ) = \ 1.9\ E7\ cm^3$

To find mass of the volume, $m = \rho\ v = (2.3\ g/cm^3\ )\ (1.9E7\ cm^3) =$ $\mathbf{4.4\ E7\ g}$

	Energy disposition (MeV)	Absorbed dose (MeV/g)
	0.001	2.20E-11
	0.005	1.10E-10
	0.01	2.30E-10
	0.1	2.30E-09
	0.5	1.10E-08
	1	2.20E-08
	3	6.80E-08
	5	1.20E-07
	7	1.60E-07
	10	2.30E-07
	13	2.90E-07
	15	3.40E-07
Average	*4.5*	1E-07

Applying the absorbed dose equation to a range of neutron energies results in a average absorbed dose of $\boxed{1.0\ E\text{-}07\ MeV/g}$

This analytical result was used to compare to the MCNP result for absorbed dose. The input file ran for 1.0E9 particles, using 1293.81 minutes of computer time.

The tally for average energy deposition 30 cm away from the concrete shield was **3.2813E-15 MeV** with an error of 0.2491. This error is not surprising, because the area being measured is relatively small. Converting this energy into absorbed dose (multiplying by the mass of concrete volume in this case) gives a average absorbed dose of 1.44 E-07 MeV/g, very close to the analytical result.

While the error is large, the value of mean energy measured stayed constant as particles were added (Figure 11). This agreement of results can further be verified by running the input file for more particles, so that error is reduced to < %5.

		tally		6	
nps	mean	error	vov	slope	fom
65536000	6.1897E-18	1.0000	1.0000	0.0	1.4E-03
131072000	1.8908E-15	0.5038	0.2670	0.0	2.7E-03
196608000	2.4291E-15	0.3594	0.1399	0.0	3.5E-03
262144000	2.5668E-15	0.3153	0.1163	0.0	3.4E-03
327680000	2.3554E-15	0.2842	0.1034	0.0	3.3E-03
393216000	2.0349E-15	0.2750	0.1022	0.0	3.0E-03
458752000	1.8598E-15	0.2628	0.0960	0.0	2.8E-03
524288000	1.7022E-15	0.2532	0.0931	0.0	2.6E-03
589824000	2.0299E-15	0.2419	0.1215	0.0	2.6E-03
655360000	2.2308E-15	0.2179	0.0949	0.0	2.8E-03
720896000	2.1545E-15	0.2076	0.0906	0.0	2.8E-03
786432000	2.5177E-15	0.2318	0.2424	0.0	2.1E-03
851968000	2.4726E-15	0.2218	0.2267	0.0	2.1E-03
917504000	2.7145E-15	0.2265	0.1944	0.0	1.9E-03
983040000	3.2003E-15	0.2584	0.3055	0.0	1.3E-03
1000000000	3.2813E-15	0.2491	0.2991	0.0	1.4E-03

Figure 11: Average energy measured outside concrete shield in MCNP

IV. DISCUSSION & CONCLUSIONS

A low power HTGR is a viable choice for a university nuclear reactor. The inherent safety features, compact size, and high flux make such a design attractive. Using the HTR-10 as a design basis validates the study, and shows an operational 10MW pebble bed reactor is feasible. The project was successful in creating a simple design for computational testing, and future work will be able to add detail and create more complex core geometry. The code was verified with a estimated calculation of absorbed dose outside the shield.

In the grand scheme, the existence of a WPI research reactor would compete with the existing university reactors in the Northeast United States, and could safely power a portion of the university facilities. The design of such infrastructure could be spread across many disciplines and projects at WPI and collaboration with outside agencies. In the short term, the next steps to the project are much more simple. The geometry in MCNP could be made more complex, accounting for the fuel pebbles, reflectors, and the beam port for experiments. Ultimately, a very advanced geometry (like in Figure 12) could be developed. Such a model accounts for the random fuel distribution of the pebbles and includes all the components of the reactor module.

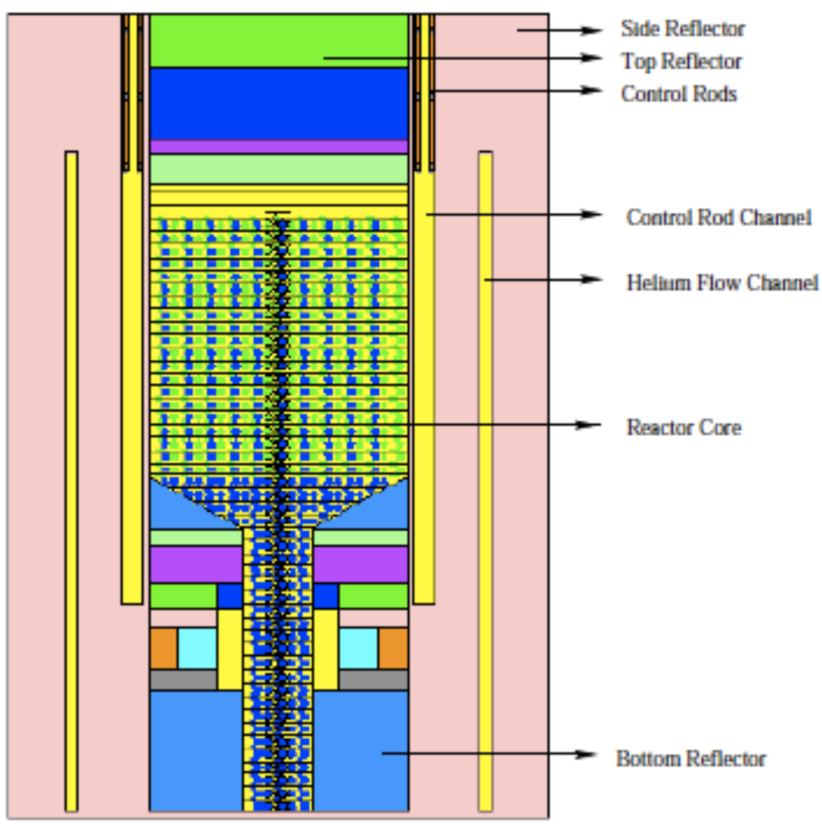


Figure 12: advanced MCNP model of the HTR-10 (Colak)

ACKNOWLEDGEMENTS

Professor David Medich, my advisor for this project, for his endless help over my years at WPI, and for his confidence and patience.

The WPI Physics Department, especially Professor Germano Iannacchione and Professor Izabela Stroe for their tutelage and support.

VII. REFERENCES

Colak, Seker; “MONTE CARLO CRITICALITY CALCULATIONS FOR A PEBBLE BED REACTOR WITH MCNP”, 2003

International Atomic Energy Agency (IAEA), “*A*dvances in Small Modular Reactor Technology Developments”, 2014, Austria

Oak Ridge National Laboratory (ORNL), “TRISO- Coated Fuel Processing to Support High Temperature Gas Cooled Reactors”, 2002

AtomicInsights, http://atomicinsights.com/advances-high-temperature-nuclear-reactor-fuel-triso-integrity-1800-c/, 2009

Reitsma, Frederick; “Pebble-bed Reactor Core Neutronics Design and Fuel Cycle”, Calvera, IAEA Course on HTGR technology, 2012

Yuanhui Xu, Shouyling Hu, Fuli, and Suyuan Yu; “HIGH TEMPERATURE REACTOR DEVELOPMENT IN CHINA”, Institute of Nuclear and New Energy Technology, Tsinghua University, 2005

Zongxin Wu, Dengcai Lin, Daxin Zhong; “The design features of the HTR-10”, Institute of Nuclear Energy and Technology, Tsinghua University, 2002

ADDITIONAL REFERENCES

Hecht, M.M.; “Modular High-Temperature Reactors Can Change the World”, 2008

H. Hidayatullah, S. Susyadi, M. Hadid Subki*“Design and technology development for small modular reactors: Safety expectations, prospects and impediments of their deployment”, Nuclear Power Technology Development Section, Division of Nuclear Power Department of Nuclear Energy, 2014

Kuijper, Raepsaet; “HTGR reactor physics and fuel cycle studies”, 2005

Liverhant, S.E., *Nuclear Reactor Physics*, 1960, Wiley & Sons

Meghreblian, Robert; *Reactor Analysis,* 1960, McGraw -Hill

Odmaa Sambuu, Toru Obara; Comparative study on HTGR designs for passive decay heat removal, Department of Nuclear Engineering, Tokyo Institute of Technology, 2013

Woodruffa, Millerb; “Cost sensitivity analysis for a 100 MWe modular power plant and fusion neutron source”, Woodruff Scientific Inc., 2014

Zerovnik, Podvratnik, Snoj; “On normalization of fluxes and reaction rates in MCNP criticality calculations”, Jozef Stefan Institute, 2013